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A. Edward Scherer
Manager of
Nuclear Regulatory Affairs

October 22, 2004

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20055

Subject: **Docket Nos. 50-361 and 50-362**
Response to Request for Information to Proposed Change Number (PCN) 534, "Containment Penetrations"
San Onofre Nuclear Generating Station Units 2 and 3

References:

1. Letter dated August 4, 2003, from Dwight E. Nunn (SCE) to Document Control Desk (NRC), Subject: Proposed Change Number (PCN) 534, "Containment Penetrations"

2. Letter dated December 24, 2003, from A. E. Scherer (SCE) to Document Control Desk (NRC), Subject: Response to Request for Additional Information (RAI) regarding Containment Structure Equipment Hatch Shield Doors

3. Letter dated June 3, 2004, from Dwight E. Nunn (SCE) to Document Control Desk (NRC), Subject: Response to Request For Additional Information and Supplement 1 to Proposed Change Number (PCN) 534, "Containment Penetrations"

4. Letter dated August 24, 2004, from A. E. Scherer (SCE) to Document Control Desk (NRC), Subject: Response to Request For Additional Information to Proposed Change Number (PCN) 534, "Containment Penetrations"

Dear Sir or Madam:

This letter provides responses to NRC questions on Proposed Change Number (PCN) 534. The Enclosure provides information as discussed with NRC Staff reviewers on September 9, 2004 and subsequent telecons.

SCE has evaluated the information in the Enclosure and concludes that there is no change to the previous finding of "no significant hazards consideration."

Once approved, the subject amendment shall be implemented within 60 days.

P.O. Box 128
San Clemente, CA 92672
949-368-7501
Fax 949-368-7575

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If you have any questions or require additional information, please contact Mr. Jack Rainsberry at 949-368-7420.

Sincerely,

Enclosure: RAI Responses

cc: B. S. Mallett, Regional Administrator, NRC Region IV
B. M. Pham, NRC Project Manager, San Onofre Units 2 and 3
C. C. Osterholtz, NRC Senior Resident Inspector, San Onofre Units 2 and 3
S. Y. Hsu, Department of Health Services, Radiologic Health Branch

ENCLOSURE

RAI RESPONSES

RAI 1

PCN-534 Table 1 indicates that the fuel handling accident inside containment (FHA-IC) dose calculation models a relative power distribution (RPD) peaking factor of 1.37 applied to 210 failed fuel rods, and a radial peaking factor (RPF) of 1.71 applied to 16 failed fuel rods. An August 24, 2004 letter responding to a request for additional information noted that the validity of the FHA-IC dose calculation RPD input parameter is verified each cycle. Discuss how the RPD and RPF are verified on a cycle-specific basis.

RESPONSE TO RAI 1:

Southern California Edison is licensed to perform fuel reload analyses for SONGS Units 2 and 3. The methodology for performing these analyses is discussed in the NRC-approved SONGS Units 2&3 Reload Topical Report (Reference 1). SONGS procedure SO23-XXXVI-2.10 (Reference 2) controls the reload analysis process. Section 1.5 of this procedure addresses the reload cycle dose analysis validation for the fuel handling accident. The tasks include verification or modification of the input parameters, including the peaking factors modeled in the current FHA-IC dose analysis.

A change to the fuel handling accident analysis was approved in-house in accordance with 10 CFR 50.59 in January 2000 and included in the Updated Final Safety Analysis Report (UFSAR) update submitted to the NRC August 2001. As described in UFSAR Chapter 15.10.7.3.9, the worst-case fuel handling accident inside containment is a damage of 16 fuel rods in a dropped fuel assembly on to 210 damaged fuel rods in an impacted assembly. For this scenario, the FHA-IC dose analysis modeled a Radial Power Density (RPD) of 1.37 applied to the impacted 210 failed fuel rods and a Radial Peaking Factor (RPF) of 1.71 applied to 16 failed fuel rods. This FHA-IC dose analysis, Calculation N-4072-003 (Reference 3), was submitted to the NRC in a December 24, 2003 letter responding to a request for additional information.

In accordance with SONGS procedure SO23-XXXVI-2.10, the cycle-specific validation task is performed in two steps. The core reload physics analysis first determines the worst-case RPF's and RPD's for the upcoming cycle. The dose verification evaluation then compares the cycle-specific physics calculated RPF's and RPD's against the input values in the FHA-IC dose analysis of record (AOR). If the cycle-specific values bound the AOR input values, then the FHA-IC dose calculation is validated for the cycle of interest. If the cycle-specific input values do not bound the AOR input values, then the dose calculation is revised and reviewed per 10 CFR 50.59 to determine if there is a need for NRC approval.

The validation of peaking factors was most recently performed for SONGS Unit 2 Cycle 13 (U2C13). The cycle-specific physics calculations that determine the RPF and RPD are determined using the NRC approved ROCS-MC computer code (see UFSAR Section 4.3.3 for details on this code methodology). NRC approval for SCE to use this code methodology is described in Sections 2.0 and 3.2.6 of the SER for the SONGS Units 2&3 Reload Topical Report.

For purposes of the FHA-IC dose analysis the RPD has been calculated for an entire fuel assembly of 236 rods instead of 210 rods as strictly needed for the dose analysis. This simplification has been performed since the approved physics code package calculates worst-case RPD on an assembly wide basis and error-likely hand manipulation would be necessary to develop an exact 210 rod worst-case RPD. In order to accommodate this simplification, the FHA-IC dose analysis has used a RPD of 1.37 and an iodine gap fraction of 12% for both high and low burnup fuel. The following values from the U2C13 analyses demonstrate the usage and conservatism in the RPD/RPF methods.

Input Parameter	**FHA-IC Dose Analysis Value (per Reference 3)**	**U2C13 Reload Analysis Value**
RPF applicable to 16 failed fuel rods	1.71	1.65
Parameters for once burned assemblies:		
RPD for 236 failed fuel rods	1.37	1.322
RPD for 210 failed fuel rods	1.37	1.334
Rod iodine release gap fraction	12%	10%
Parameters for >once burned assemblies:		
RPD for 236 failed fuel rods	1.37	1.130
RPD for 210 failed fuel rods	1.37	1.137
Rod iodine release gap fraction	12%	12%

RAI 1 REFERENCES:

1. Document SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3", June 1999

2. Procedure SO23-XXXVI-2.10, "Core Reload Analyses and Activities Checklist"

3. Calculation N-4072-003, "Fuel Handling Accident (FHA) Inside Containment – Control Room & Offsite Doses"

4. UFSAR 2/3-15.10 Change 15

RAI 2

SCE stated that the maximum RPD is calculated for an entire fuel assembly of 236 rods instead of 210 rods as strictly needed for the dose analysis. SCE indicated that this simplification was performed because the physics code package calculates the assembly-wide maximum RPD and error-likely hand manipulation would be necessary to develop an exact 210-rod worst-case RPD. However, the staff is concerned that the assembly-wide 236-rod maximum RPD may not bound the 210-rod maximum RPD in some circumstances. The staff requests the licensee provide additional information to demonstrate how it determines the maximum bounding 210-rod RPD each cycle.

RESPONSE TO RAI 2:

The RPD characterizing 210 fuel rods will be greater than or equal to the RPD characterizing all 236 fuel rods in one assembly. However, the difference between the RPD values is small (i.e., less than one percent). The FHA-IC dose analysis includes margin by modeling an RPD that bounds both the low and high burnup cycle-specific RPD's, and by conservatively modeling a 12 percent iodine release gap fraction for both low and high burnup fuel. As discussed in the response to RAI 1, low burnup fuel is characterized by a 10 percent iodine release gap fraction. The combined effect of using a maximum RPD for 236 pins (independent of burnup) coupled with the maximum iodine release gap fraction is conservative relative to an RPD for 210 pins and the corresponding iodine release gap fraction. The determination of the RPD uses the approved methodology as discussed in the response to RAI 1.

RAI 3

Since the 210-rod RPD is non-conservative when compared to the 236-rod RPD used in the reload analysis, the staff requests that for high burnup fuel, SONGS consider placing a corresponding and appropriate bias on the reload analyses to ensure that the FHA analysis RPD remains bounding for all future reloads.

RESPONSE TO RAI 3:

Future SONGS reload analyses will no longer calculate a maximum 236-pin RPD for the FHA analysis. SONGS will calculate a cycle-specific maximum 210-rod RPD and use the corresponding fuel rod iodine gap release fraction for that assembly. This product will be compared with the product of the RPD and fuel rod iodine release fraction modeled in the FHA dose analysis of record to ensure that the FHA analysis remains bounding for all future reloads.

RAI 4

SCE's reload analysis assumes a specific depletion pattern over the entire operating cycle. However, numerous anticipated operating conditions can perturb the flux profile in the core resulting in higher localized peaking factors. The staff requests SCE provide additional information to describe how core operating conditions not analyzed as part of the reload analysis are accounted for in determining the maximum RPF and RPD do not exceed the FHA-IC assumptions during the cycle.

RESPONSE TO RAI 4:

The calculation of the RPD and RPF are per the methodology described in the NRC-approved SONGS Units 2&3 Reload Topical Report (Reference 1). Section 3.1.1 of the topical states:

> "Two design depletions are performed at Hot Full Power (HFP) based on two different points within the previous cycle shutdown window, short (best estimate cycle Effective Full Power Days (EFPD) minus delta) and long (best estimate cycle EFPD plus delta). These depletions supply core isotopic distributions and nominal HFP power distributions, which bound any possible nominal distributions of the reload fuel cycle as long as the previous cycle ends between the short and long end points."

The methods used to determine radial peaking factors and radial power density are also described in the San Onofre 2&3 UFSAR Section 4.3.2. Specifically, Section 4.3.2.2.3 states:

> "Figures 4.3-2 through 4.3-10 and 4.3-11 through 4.3-13 show typical planar radial and unrodded core average axial power distributions, respectively. They illustrate conditions expected at full power for various times in the fuel cycle as specified on the figures. It is expected that the normal operation of the reactor will be with limited CEA insertion so that these power distributions represent the expected power distribution during most of the cycle."

In addition, UFSAR Section 4.3.2.5 states:

> "It is expected that the core will be essentially unrodded during full power steady-state operation, except for limited insertion of the first regulating group in order to compensate for minor variations in moderator temperature and boron concentration as described in paragraph 4.3.2.4.2."

Further discussion of core radial and axial stability is also provided in UFSAR Section 4.3.2.7.3.

Therefore, the reload analysis accounts for all operating conditions required to evaluate peaking factors for the FHA-IC dose analysis.

RAI 4 REFERENCE:

1. Document SCE-9801-P-A, "Reload Analysis Methodology for the San Onofre Nuclear Generating Station Units 2 and 3", June 1999

RAI 5

Explain how the SONGS Units 2&3 reload methodology for calculating relative power densities (RPD's) accounts for core changes just prior to shutdown, and its effect on the fuel handling accident inside containment (FHA-IC) event consequences.

RAI 5 RESPONSE:

The fuel handling accident inside containment (FHA-IC) source term present in the fuel rod gap spaces and available for release is the product of the number of failed fuel rods (210 or 16, as appropriate), the FHA activity inventory of an average fuel rod, the peaking factor (RPD or RPF, as appropriate), and the fuel rod gap isotopic inventory fractions. SONGS UFSAR Section 15.10B.2.2.2 presents the FHA activity inventory for a single fuel rod. The single fuel rod FHA activity inventory reflects operation at rod average power based on the maximum core power, currently 3458 MWt, to the maximum fuel pin burnup limit, currently 60 GWD/MTU.

SONGS Units 2&3 reload methodology calculates RPD's using nominal depletions as described in the response to RAI 4. The cycle maximum RPD value of these nominal depletions is used in the calculation of the FHA-IC source term. The use of the cycle maximum RPD for the FHA-IC source term conservatively assumes that the cycle maximum RPD had existed for the entire power operation up to the maximum fuel pin burnup. As a result, the FHA-IC source term methodology is not affected by short-term core changes that occur just prior to shutdown.

Therefore, the reload analysis conservatively calculates RPD's for the FHA-IC dose analysis for all operating conditions.

RAI 6

Describe the configuration of the ductwork that is the suspected inleakage pathway.

RESPONSE TO RAI 6:

The Control Room Cabinet Areas for each of Units 2 and 3 are served by a Normal A/C Unit, a Train A Emergency A/C Unit, and a Train B Emergency A/C Unit. On each generating unit, the Train A Emergency A/C Unit and Normal A/C Unit share a common discharge duct, but have independent suction ducts. Each suction duct is equipped with a pneumatic motor operated damper. The discharge duct of the Normal A/C Unit is provided with a backdraft damper. For each generating unit, the Train B Emergency A/C Unit has independent suction and discharge ductwork; the discharge duct is provided with a motor operated damper.

All suction ductwork leaves the Control Room Cabinet Areas at approximately 43' elevation and is drawn up to the fan rooms on the 50' level, outside the (Control Room Envelope) CRE. The air passes through the fan and cooling coils before being discharged back to the Cabinet Areas. There are flexible duct fittings at the suction and discharge of each fan. The suction side of the fan has relatively low pressure compared to the mechanical room, which is the likely source of the Control Room Envelope in-leakage, as described in the response to RAI 8. The Normal A/C unit casings are not seam welded, but have been tested and shown to be leak tight.

RAI 7

Provide data on pressure differences from inside the control room envelope and adjacent areas.

RESPONSE TO RAI 7:

During each of the single-train emergency mode CRE inleakage tests, Delta P data was taken at more than 30 different locations. Provided in the following table (in inches of water) are minimum, maximum, and average pressures, as well as the data points for the fan rooms in which potential inleakage was indicated.

Control Room Inleakage Testing Delta-P Data
(Inches of Water)

Test	Minimum Delta-P	Maximum Delta-P	Average Delta-P	Fan Room Delta-P
Train A	0.546	1.059	0.820	Room 309A 0.890 Room 309D 0.889
Train B	0.605	0.974	0.788	Room 309C 0.844

RAI 8

What is the basis for concluding that 132 cfm (with uncertainty, 259 cfm) is the maximum expected control room envelope inleakage during dual train emergency mode operation? Could dual train operation pressurize an area adjacent to the control room envelope such that a leakage path is worsened or created?

RESPONSE TO RAI 8:

The CRE Inleakage Testing Report estimated the amount of inleakage into control room cabinet cooler air handling unit system using tracer gas concentrations upstream and downstream of the duct. For some sections of ducting the concentration ratios are not statistically different from unity, implying that no inleakage was measured along the negative differential pressure segments of the cabinet cooler system(s) ductwork. For other sections, the duct inleakage calculations yielded inleakage rates that were similar to the total CRE inleakage rate. The test results suggest that nearly all of the unfiltered inleakage may originate from the low-pressure ducting associated with the A and B Train control room cabinet area coolers.

The delta-P data discussed in the response to RAI 7 supports the conclusion that the source of inleakage is from the suction side of the cabinet coolers; the CRE is pressurized relative to all adjacent areas on 9', 30' and 50', thereby precluding inleakage from these areas. It is also worth noting that the CREACUS units are entirely contained within the CRE, and are therefore not subject to a similar suction-side inleakage scenario. Also, ductwork that completely traverses the envelope, between 9' and 50' is of welded construction, and has been smoke tested and found to be leak-tight.

During each single-train emergency mode CRE inleakage test, a nominal outside air flow of approximately 2000 cfm is brought into the CRE, through the associated emergency ventilation unit. In order to achieve a flow balance, the differential pressures between the control room and adjacent areas rise to the values cited above, such that the 2000 cfm inflow is driven out of the CRE to the adjoining areas.

In dual-train emergency mode operation, both emergency ventilation units operate, increasing the outside air flow to the CRE by approximately 1600 cfm. (The full nominal additional flow of 2000 cfm is not achieved, as with the rise in control room pressure, the operating point on ventilation units' fan curve shifts). With a combined outside air flow of approximately 3600 cfm, the pressure in the control room will rise to a pressure higher than that experienced in single-train operation, in order to achieve the flow balance between the outside air being brought in by the ventilation units, and the outleakage to the adjoining spaces. Simply put, higher outleakage flows require higher driving differential pressures.

For either single-train or dual-train operation, the 2000 cfm or 3600 cfm outleakage (respectively) from the CRE to adjoining spaces will have negligible effect on the pressures in these adjoining spaces. The adjoining spaces are served by separate HVAC systems, which have a total circulating air flow greater than 100,000 cfm.

During each single-train emergency mode CRE inleakage test, the delta-P between the negative pressure ducting for the cabinet coolers and the fan room was calculated to be approximately -0.35" H20, which would induce the measured inleakage. This value is due to the negative pressure created upstream of the fan intake; however, it is dependent on the delta-P between the CRE and the fan room, as the duct draws suction from the CRE. Dual-train operation would not increase this negative delta-P; rather, as discussed above, dual train operation would result in a higher delta-P between the CRE and adjoining spaces - including the fan room. This would increase the pressure within the ducting. This would, in turn, reduce the negative delta-P between the ducting and the fan room, reducing the amount of inleakage. In fact, the increase in CRE pressure could potentially result in a positive delta-P between the suction ductwork and the fan room, thereby eliminating the inleakage.

It is therefore concluded that the inleakage during dual train operation is bounded by the sum of the Train A inleakage (67 cfm) and the Train B inleakage (65 cfm), or 132 cfm. This number is a bounding value, because the increased Control Room pressurization due to dual train operation would reduce or eliminate the inleakage.

RAI 9

Provide a comparison of the maximum expected inleakage during dual train operation to the amount of inleakage assumed in analyses supporting the License Amendment Request.

RESPONSE TO RAI 9:

The design basis fuel handling accident inside containment radiological analyses to support PCN 543 assumed a value of 990 cfm unfiltered inleakage into the Control Room Envelope; an additional 10 cfm was assumed for CRE ingress and egress. The 990 cfm boundary inleakage rate is nearly 4 times the value of the maximum inleakage for dual train operation of the CREACUS in the emergency mode (259 cfm, including test uncertainty).